BOSTON EDISON COMPANY
800 BOYLSTON STREET
BOSTON, MASSACHUSETTS 02199

WILLIAM D. HARRINGTON
SENIOR VICE PRESIDENT
NUCLEAR

February 21, 1984
BECo 84-029

Mr. Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

License No. DPR-35
Docket No. 50-293

Revi-ion 1 to Reload 6 Licensing Submittal

Dear Sir:

The General Electric Supplemental Reload Licensing Submittal for the Pilgrim Nuclear Power Station was submitted by letter from W. D. Harrington to D. B. Vassallo, dated December 28, 1983 (BECo 83-301). In this letter, we identified an error that was noted on Page 13 of the reload licensing submittal. This error involved the upper right hand graph in Figure 5, Plant Response to MSIV Closure, 100% Power, 107.5% Flow (Page 13). Specifically, Curve 3 of this graph was labeled the relief valve flow curve, but it actually represented the sum of safety and relief valve flows.

The attached revision to this reload licensing submittal corrects the error noted on Page 13 and is submitted for your use. Correction of this error does not affect the conclusions of the reload licensing submittal.

Very truly yours,

W D Harrington

DMV/kmc

Attachment: General Electric Boiling Water Reactor Supplemental Reload Licensing Submittal for Pilgrim Nuclear Power Station Unit 1, Reload 6, 22A1694, Revision 1, December 1983 (40 copies).

IE26
Add: Reg File

22A1694
REVISION 1
CLASS I
DECEMBER 1983

GENERAL ELECTRIC BOILING WATER REACTOR SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR PILGRIM NUCLEAR POWER STATION UNIT 1, RELOAD 6

Dupe

~~8401090556~~

GENERAL ELECTRIC

22A1694
Revision 1
Class I
December 1983

GENERAL ELECTRIC BOILING WATER REACTOR SUPPLEMENTAL RELOAD LICENSING SUBMITTAL

FOR

PILGRIM NUCLEAR POWER STATION, UNIT 1 RELOAD 6

Prepared: P.E. Elliott for
P. A. Verbryke

Verified: J. S. Charnley for
C. L. Hilf

Approved: J. S. Charnley
J. S. Charnley, Manager
Fuel Licensing

NUCLEAR POWER SYSTEMS DIVISION • GENERAL ELECTRIC COMPANY
SAN JOSE, CALIFORNIA 95125

GENERAL ELECTRIC

IMPORTANT NOTICE REGARDING
CONTENTS OF THIS REPORT
PLEASE READ CAREFULLY

This report was prepared by General Electric solely for Boston Edison Company (BECo) for BECo's use with the U.S. Nuclear Regulatory Commission (USNRC) for amending BECO's operating license of the Pilgrim Nuclear Power Station. The information contained in this report is believed by General Electric to be an accurate and true representation of the facts known, obtained or provided to General Electric at the time this report was prepared.

The only undertakings of the General Electric Company respecting information in this document are contained in the contract between Boston Edison Company and General Electric Company for reload fuel fabrication for the nuclear system for Pilgrim Nuclear Power Station, dated July 14, 1972, and nothing contained in this document shall be construed as changing said contract. The use of this information except as defined by said contract, or for any purpose other than that for which it is intended, is not authorized; and with respect to any such unauthorized use, neither General Electric Company nor any of the contributors to this document makes any representation or warranty (express or implied) as to the completeness, accuracy or usefulness of the information contained in this document or that such use of such information may not infringe privately owned rights; nor do they assume any responsibility for liability or damage of any kind which may result from such use of such information.

1. PLANT UNIQUE ITEMS (1.0)*

Increased Core Flow Throughout Cycle:	Appendix A
Bounding Control Rod Drop Accident Analysis:	Appendix B
Safety/Relief Valve Low Setpoint:	Appendix C

2. RELOAD FUEL BUNDLES (1.0, 2.7, 3.3.1 AND 4.0)

	Fuel Designation	Cycle Loaded	Number	Number Drilled
Irradiated	8DB219L	4	24	24
	8DB219H	4	8	8
	P8DRB265L	5	120	120
	P8DRB282	5	64	64
	P8DRB265H	6	60	60
	P8DRB282	6	112	112
New	P8DRB282	7	160	160
	BP8DRB282	7	32	32
Total			580	580

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle core average exposure at end of cycle:	16474 MWd/ST
Minimum previous cycle core average exposure at end of cycle from cold shutdown considerations:	16074 MWd/ST
Assumed reload cycle core average exposure at end of cycle:	18070 MWd/ST
Core loading pattern:	Figure 1

*( ) refers to areas of discussion in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-6 and NEDO-24011-P-A-6-US.

4. CALCULATED CORE EFFECTIVE MULTIPLICATION AND CONTROL SYSTEM WORTH - NO VOIDS, 20°C (3.3.2.1.1 AND 3.3.2.1.2)

Minimum Shutdown Margin, BOC, k_{eff}	
Uncontrolled	1.113
Fully Controlled	0.955
Strongest Control Rod Out	0.983
R, Maximum Increase in Cold Core Reactivity with Exposure into Cycle, Δk	0.007

5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

ppm	Shutdown Margin (Δk) (20°C, Xenon Free)
700	0.048

6. RELOAD-UNIQUE TRANSIENT ANALYSIS INPUT (3.3.2.1.5 AND S.2.2)

(REDY Events Only)

	BOC7 + 6000 MWd/ST	EOC7
Void Fraction (%)	35.7	35.7
Average Fuel Temperature (°F)	1150	1150
Void Coefficient N/A* (¢/% Rg)	-6.43/-8.04	-5.65/-7.07
Doppler Coefficient N/A (¢/°F)	-0.216/-0.205	-0.228/-0.217
Scram Worth N/A ($)**		

*N = Nuclear Input Data, A = Used in Transient Analysis

**Generic, exposure independent values are used as given in "General Electric Application for Reactor Fuel," NEDE-24011-P-A-6-US.

7. RELOAD-UNIQUE GETAB TRANSIENT ANALYSIS INITIAL CONDITION PARAMETER (S.2.2)

Fuel Design	Peaking Factors (Local, Radial, Axial)			R-Factor	Bundle Power (MWt)	Bundle Flow (1000 lb/hr)	Initial MCPR
BOC7 + 6000 MWd/ST							
BP8x8R/ P8x8R	1.20	1.73	1.40	1.051	5.836	106.2	1.34
8x8	1.22	1.63	1.40	1.098	5.480	105.5	1.30
EOC7							
BP8x8R/ P8x8R	1.20	1.67	1.40	1.051	5.634	107.4	1.39
8x8	1.22	1.56	1.40	1.098	5.246	107.1	1.37

8. SELECTED MARGIN IMPROVEMENT OPTIONS (S.2.2.2)

Transient Recategorization:	No
Recirculation Pump Trip:	No
Rod Withdrawal Limiter:	No
Thermal Power Monitor:	No
Measured Scram Time:	No
Number of Exposure Points:	2

9. OPERATING FLEXIBILITY OPTIONS

Single Loop Operation:	No
Load Line Limit:	No
Extended Load Line Limit:	Yes
Increased Core Flow:	Yes
Flow Point Analyzed:	107.5%
Feedwater Temperature Reduction:	No

10. CORE-WIDE TRANSIENT ANALYSIS RESULTS (S.2.2.1)

Transient	Flux (% NBR)	Q/A (% NBR)	ΔCPR BP8x8R/ P8x8R	ΔCPR 8x8	Figure
Exposure: BOC7 to BOC7 + 6000 MWd/ST					
Load Rejection Without Bypass	527	121	0.27	0.23	2a
Feedwater Controller Failure	318	121	0.22	0.20	3a
Exposure: BOC7 + 6000 MWd/ST to EOC7					
Load Rejection Without Bypass	580	124	0.32	0.30	2b
Feedwater Controller Failure	336	124	0.28	0.26	3b
Exposure: BOC7 to EOC7					
Inadvertent Startup of HPCI	121	114	0.13	0.13	4

11. LOCAL ROD WITHDRAWAL ERROR (WITH LIMITING INSTRUMENT FAILURE) TRANSIENT SUMMARY (S.2.2.1)

(Generic Bounding Analysis Results)

Rod Block Reading (%)	ΔCPR (All Fuel Types)
104.	0.13
105.	0.16
106.	0.19
107.	0.22
108.	0.28
109.	0.32
110.	0.36

Setpoint Selected is: 107.

12. CYCLE MCPR VALUES (S.2.2)

Nonpressurization Events:

Exposure Range: BOC7 to EOC7

	BP8x8R/ P8x8R	8x8
Inadvertent Startup of HPCI	1.20*	1.20*
Fuel Loading Error	1.24	----
Rod Withdrawal Error	1.29	1.29

Pressurization Events:

	Option A		Option B	
	BP8x8R/ P8x8R	8x8	BP8x8R/ P8x8R	8x8
Exposure Range: BOC7 to BOC7 + 6000 MWd/ST				
Load Rejection Without Bypass	1.40	1.36		
Feedwater Controller Failure	1.35	1.33		
Exposure Range: BOC7 + 6000 MWd/ST to EOC7				
Load Rejection Without Bypass	1.45	1.43	1.40	1.38
Feedwater Controller Failure	1.41	1.39	1.32	1.30

13. OVERPRESSURIZATION ANALYSIS SUMMARY (S.2.3)

Transient	P_{sl} (psig)	P_v (psig)	Plant Response
MSIV Closure (Flux Scram)	1315	1330	Figure 5

*The minimum MCPR value required by the ECCS analysis is 1.24.

14. STABILITY ANALYSIS RESULTS (S.2.4)

Rod Line Analyzed: Extrapolated Rod Block	
Decay Ratio:	Figure 6
Reactor Core Stability Decay Ratio, x_2/x_0:	0.65
Channel Hydrodynamic Performance Decay Ratio, x_2/x_0	
8x8 Channel:	0.23
BP8x8R/P8x8R Channel:	0.18

15. LOADING ERROR RESULTS (S.2.5.4)

Variable Water Gap Misoriented Bundle Analysis: Yes

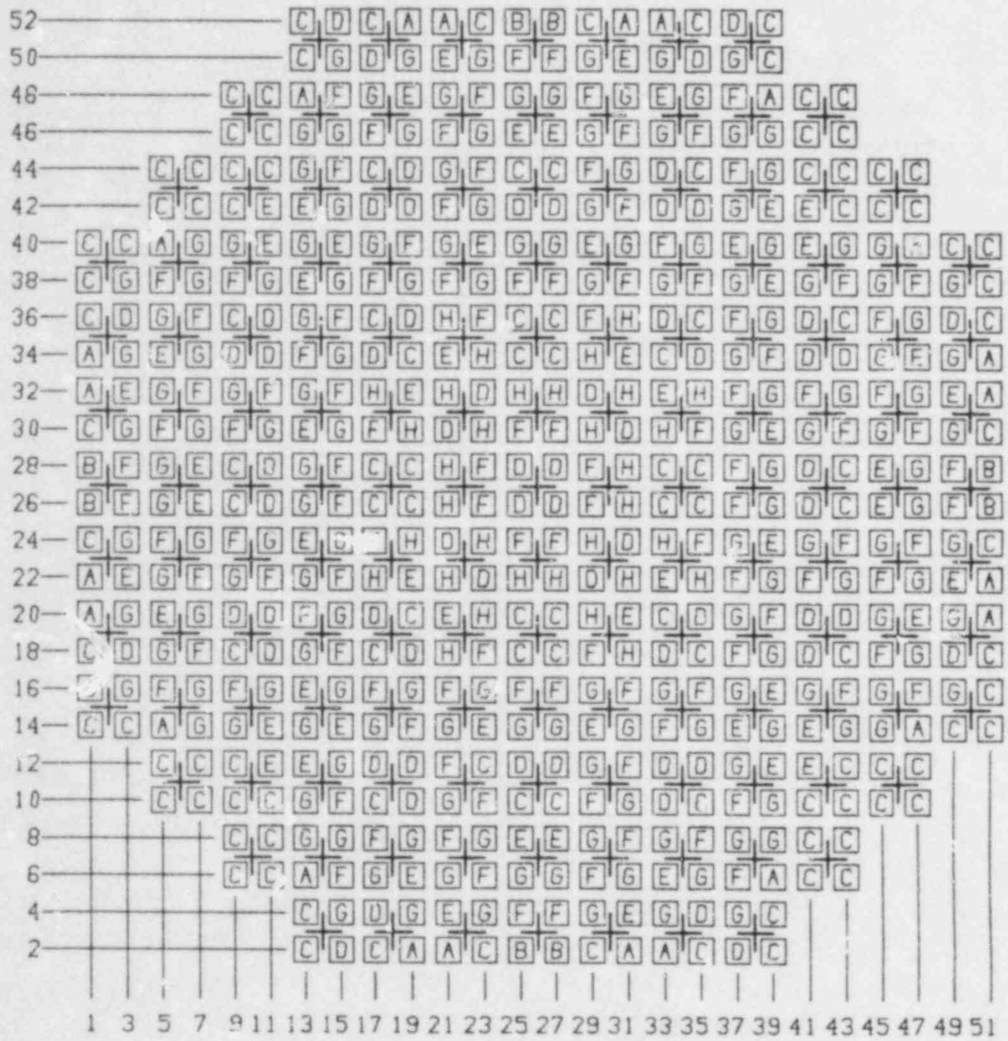
Event	Initial MCPR	Resulting MCPR
Misoriented	1.22	1.07

16. CONTROL ROD DROP ANALYSIS RESULTS (S.2.5.1)

See Appendix B.

17. LOSS-OF-COOLANT ACCIDENT RESULTS, NEW FUEL (S.2.5.2)

See "Loss-of-Coolant Accident Analysis Report for Pilgrim Nuclear Power Station," August 1977, NEDO-21696, as amended.



FUEL TYPE	
A = 8DB219L B = 8DB219H C = P8DRB265L D = P8DRB282	E = P8DRB265H F = P8DRB282 G = P8DRB282 H = BP8DRB282

Figure 1. Reference Core Loading Pattern

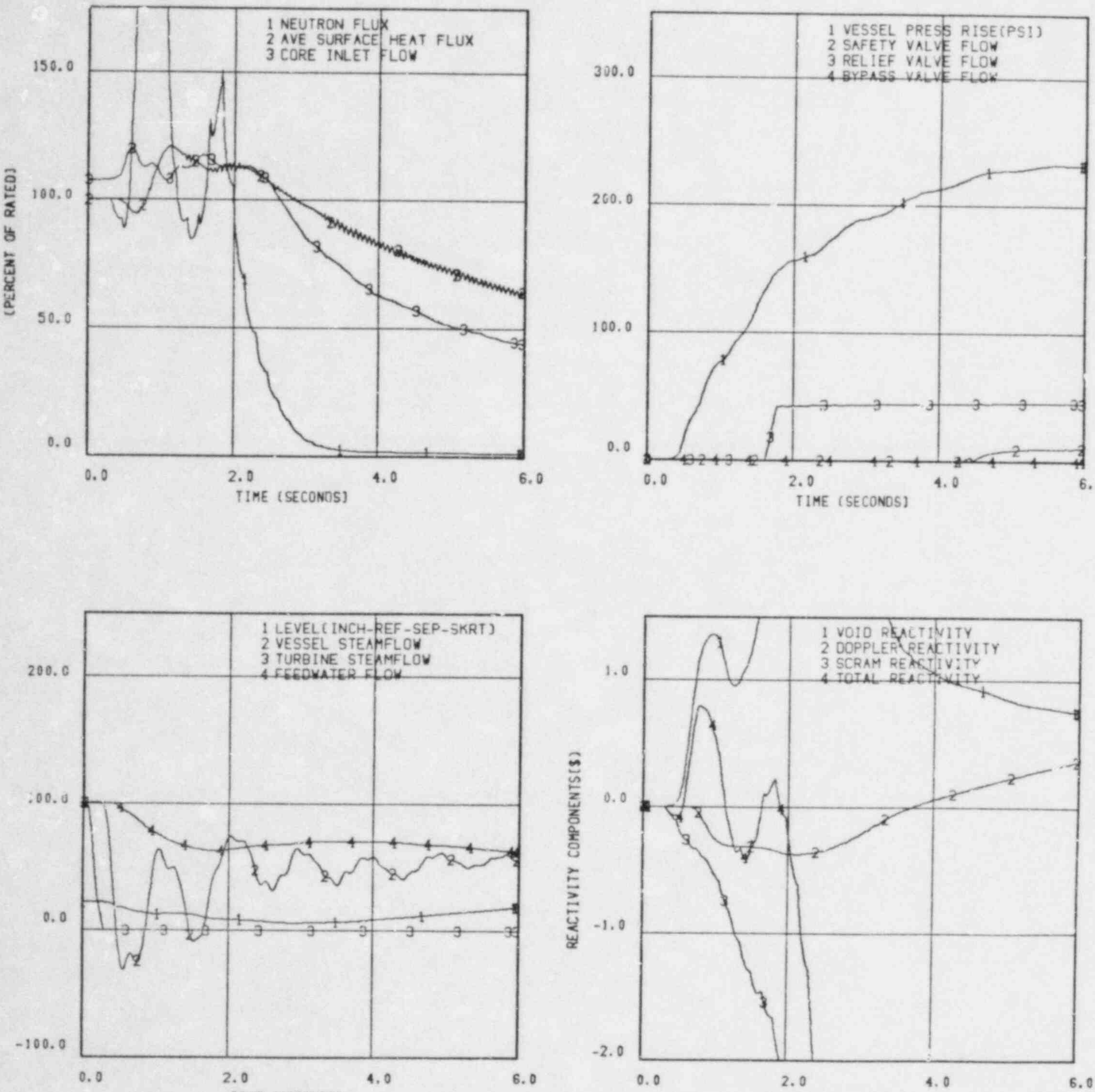


Figure 2a. Plant Response to Load Rejection Without Bypass, 100% Power, 107.5% Flow at BOC + 6000 MWd/ST

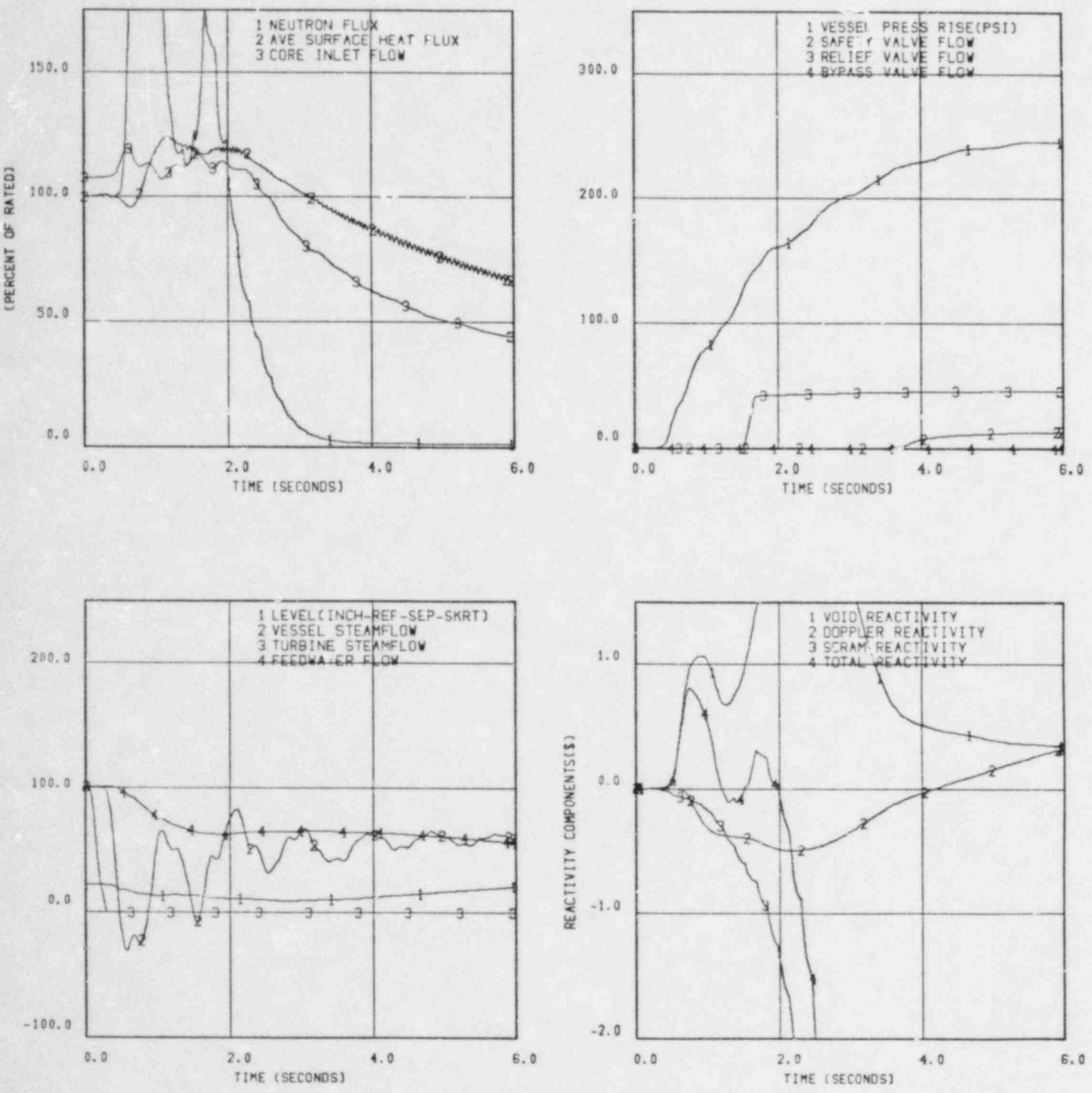


Figure 2b. Plant Response to Load Rejection Without Bypass, 100% Power, 107.5% Flow at EOC

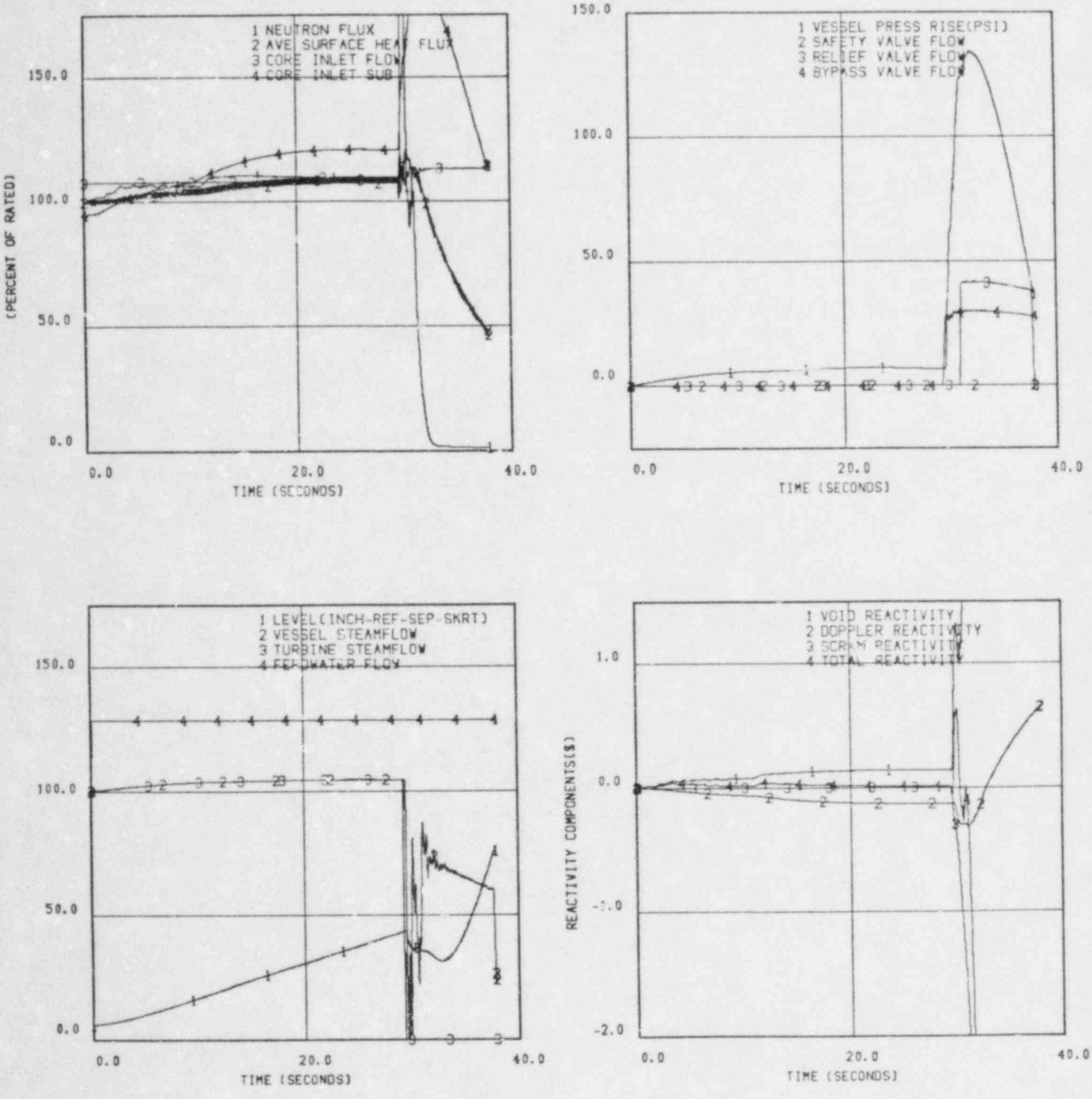


Figure 3a. Plant Response to Feedwater Controller Failure, 100% Power, 107.5% Flow at BOC + 6000 MWd/ST

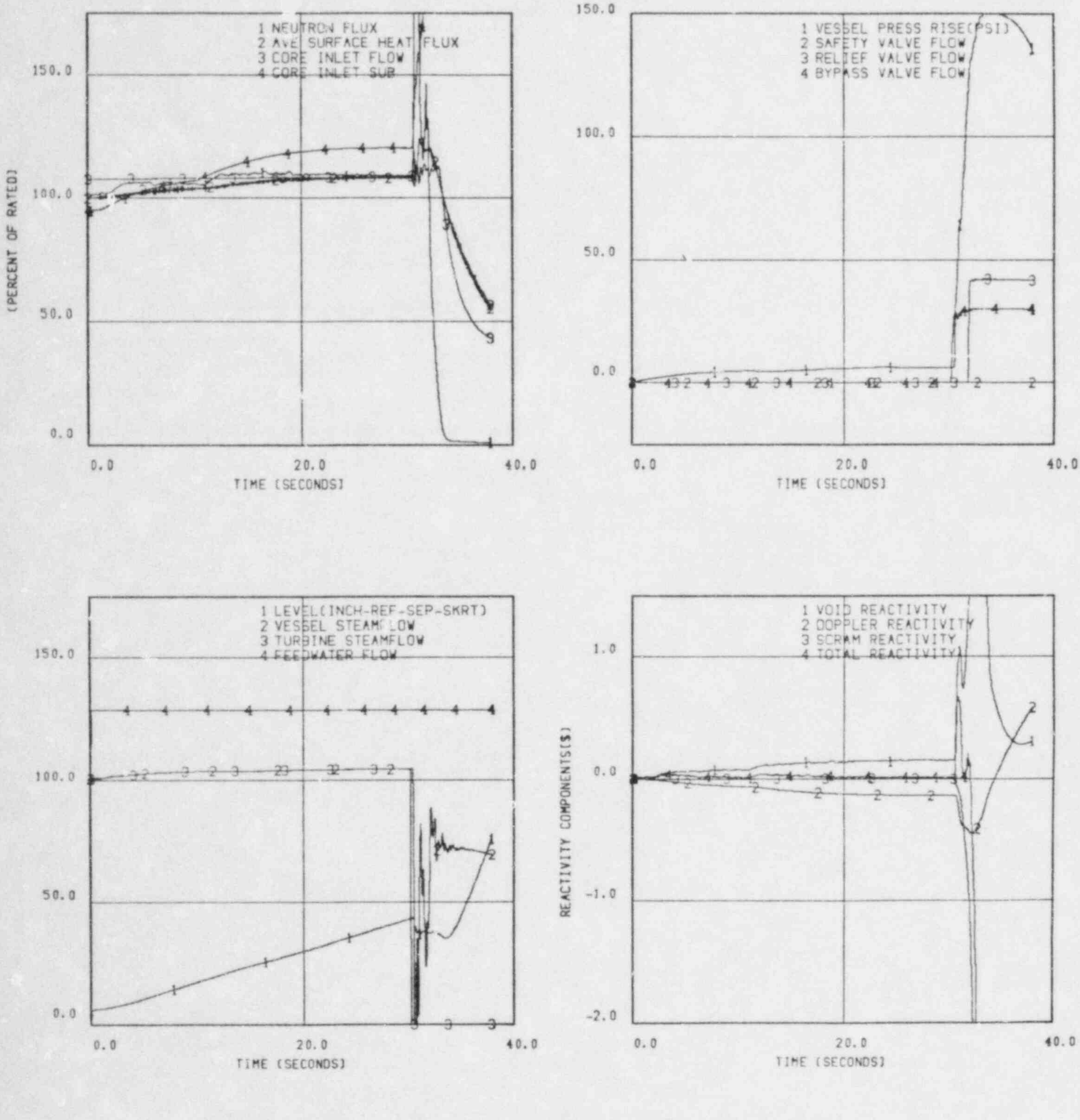


Figure 3b. Plant Response to Feedwater Controller Failure, 100% Power, 107.5% Flow at EOC

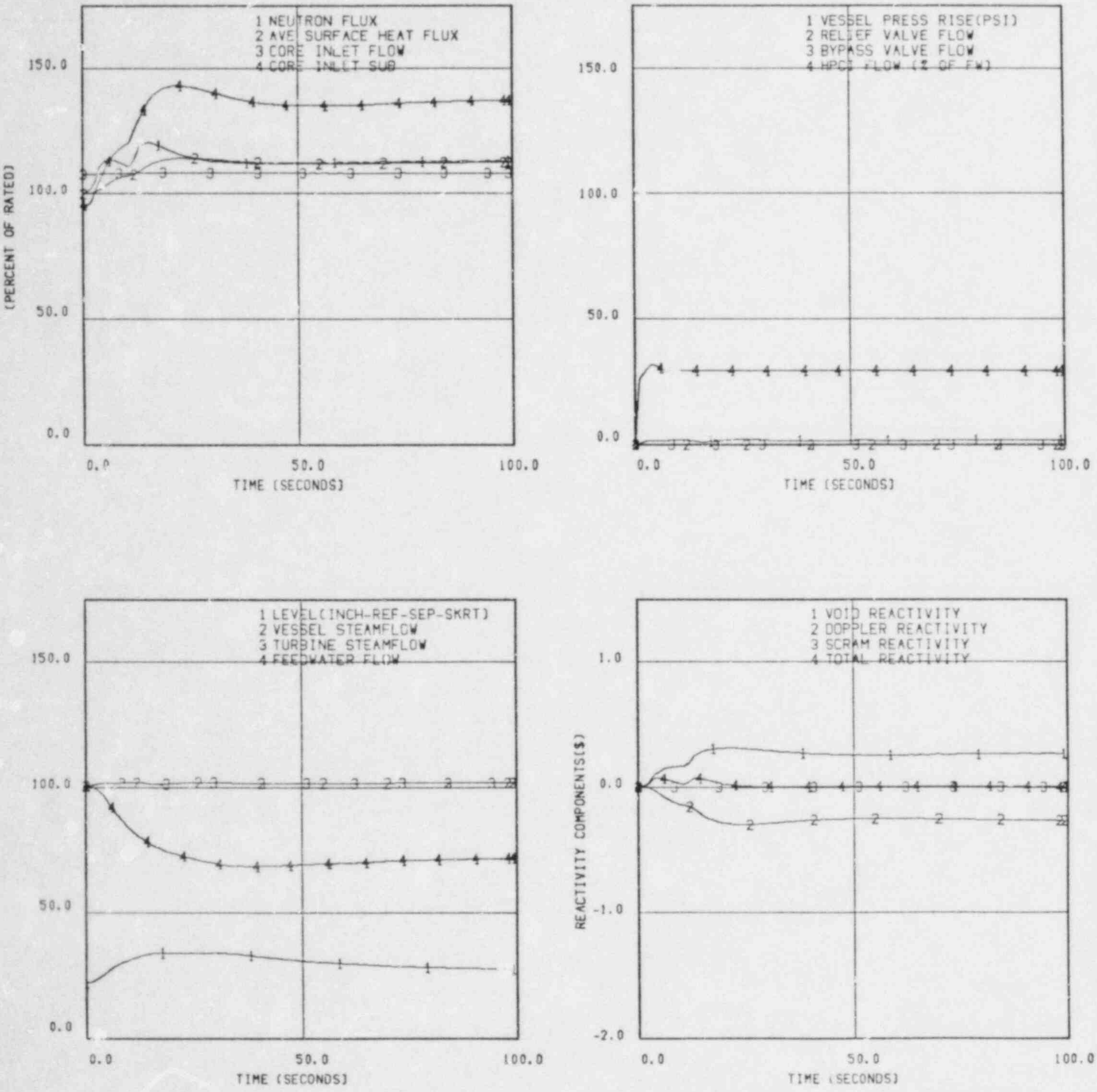


Figure 4. Plant Response to Inadvertent Start Up of HPCI Pump, 100% Power, 107.5% Flow at EOC

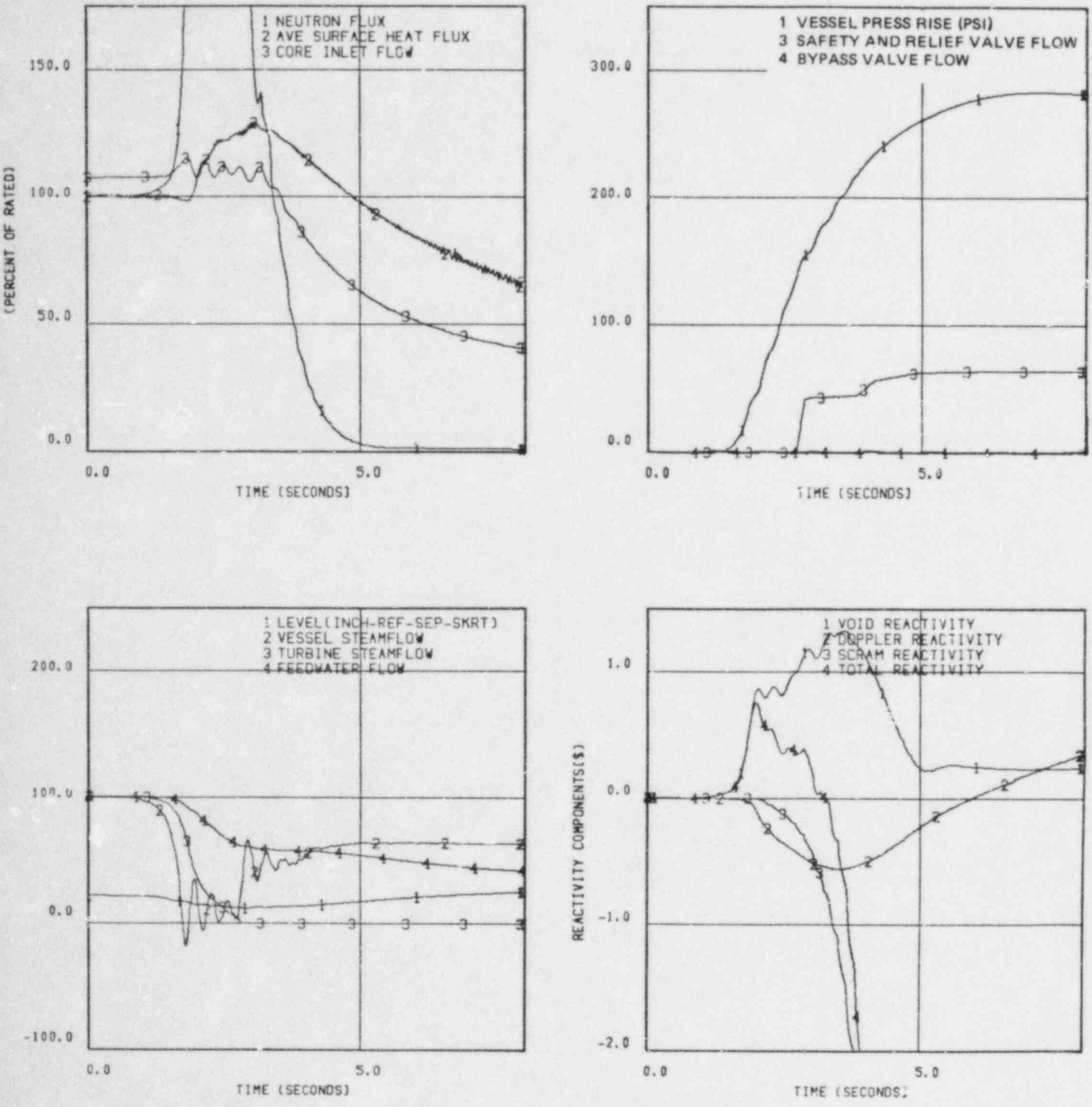


Figure 5. Plant Response to MSIV Closure, 100% Power, 107.5% Flow

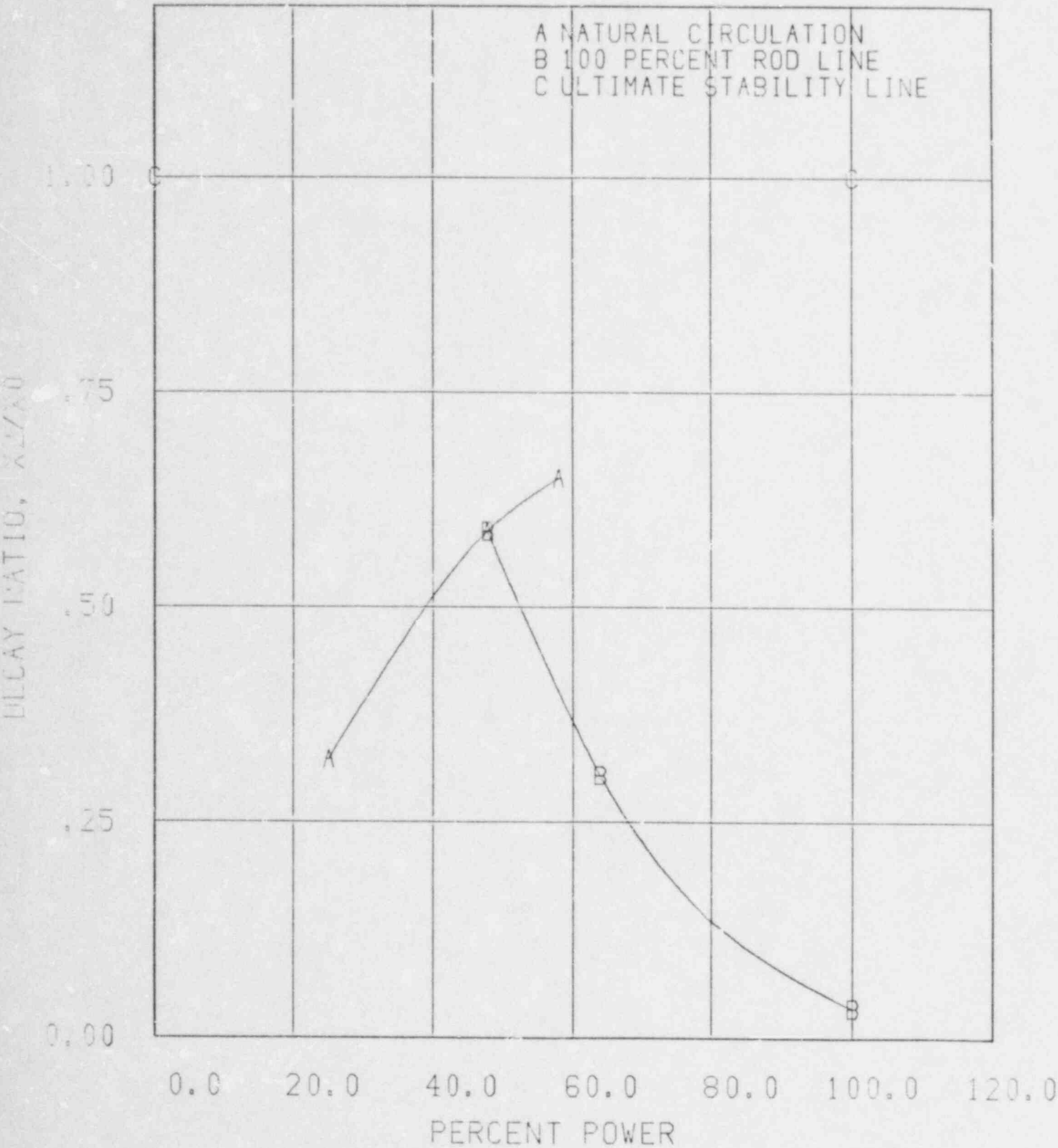


Figure 6. Reactor Core Decay Ratio

APPENDIX A
INCREASE CORE FLOW THROUGHOUT CYCLE

The analyses performed for Cycle 7 included increased core flow throughout the cycle. There are no concerns regarding reactor internals pressure drop or flow-induced vibration as discussed in the increased core flow analysis document for the EOC-6 (NEDO-30242).

The flow-biased instrumentation for the rod block monitor should be signal clipped for a setpoint of 107%, since flow rates higher than rated would otherwise result in a ΔCPR higher than reported for the rod withdrawal error.

APPENDIX B
CONTROL ROD DROP ANALYSIS

The cycle-specific control rod drop accident analysis has been discontinued for banked position withdrawal sequence (BPWS) plants based on the fact that in all cases the peak fuel enthalpy from a control rod drop accident would be much less than the 280 cal/gm limit. This change in procedures was reported and justified in Reference B-1. Reference B-2 indicates this change is acceptable to the NRC.

REFERENCES

B-1. Letter, R. E. Engel (GE) to D. B. Vassallo (NRC), "Control Rod Drop Accident," February 24, 1982.

B-2. NRC Memo, L. S. Rubenstein to G. C. Lainas, "Changes in GE Analysis of the Control Rod Drop Accident for Plant Reloads (TACS-48058)," February 15, 1983.

APPENDIX C
SAFETY/RELIEF VALVE LOW SETPOINT

The value used in the transient analyses for the safety/relief valve low setpoint is 1126 psig. This is not consistent with the value of 1106 psig reported in NEDO-24011-P-A-6-US.

GENERAL ELECTRIC